NON-PUBLIC?: N
ACCESSION #: 9006110089

LICENSEE EVENT REPORT (LER)

FACILITY NAME: McGuire Nuclear Station, Unit 1 PAGE: 1 OF 8

DOCKET NUMBER: 05000369

TITLE: Reactor Trip Occurred Because of a Failed Universal Board in the Solid State Protection System Cabinet Train "A"
EVENT DATE: 08/26/89 LER #: 89-022-01 REPORT DATE: 09/25/89

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Alan Sipe, Chairman, McGuire Safety Review Group TELEPHONE: (704) 875-4183

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: JC COMPONENT: CBD MANUFACTURER: W120
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On August 26, 1989 at 0934, a Unit 1 Reactor Trip occurred because of a Reactor Coolant (NC) low flow signal that is interlocked with a permissive signal which blocks a Reactor Trip when the unit is less than 48 percent power (P-8 permissive). Unit 1 was operating in Mode 1, Power Operation, at 100 percent power prior to the trip. A low NC flow signal with P-8 permissive caused Reactor Trip Breaker "A" to trip the unit. The signal was caused by a failed Universal Board in the Solid State Protection System (SSPS) cabinet for Train "A". The Turbine Generator automatically tripped because of the Reactor Trip. All systems and equipment responded as expected following the trip with one exception. Operations personnel implemented the Reactor Trip recovery procedure to recover from the transient. At about 1000, Operations personnel made the required notification to the NRC. At 2220, Instrumentation and

Electrical personnel discovered the failed Universal Board and replaced it. The SSPS Cabinet was tested after the board was replaced to ensure the train would operate properly. Unit 1 was returned to Mode 1, Power Operation, on August 28, 1989 at 1230. This event is assigned a cause of Equipment Failure / Malfunction. This event is Nuclear Plant Reliability Data System reportable. The board will be sent to Westinghouse for repair and failure analysis.

END OF ABSTRACT

EVALUATION:

Background

The Solid State Protection System (SSPS) takes digital inputs (voltage/no voltage) from the 7300 Process Control System and nuclear instrument channels EIIS:CHA! corresponding to conditions (normal/abnormal) of unit parameters. The system combines these signals in the required logic combination and generates a trip signal (no voltage) to the undervoltage coils of the Reactor EIIS:RCT! Trip circuit breakers EIIS:52! when the necessary combination of signals occur. The system also provides annunciator EIIS:ANN! status lights EIIS: IL! and computer EIIS: CPU! input signals which indicate the condition of bistable input signals, partial trip and full trip functions and the status of the various blocking, permissive EIIS:69! and actuation functions. Bistables actuate at the setpoints at which a signal would be initiated through the Reactor Protection System (RPS) EIIS:JC!.

Annunciation for Reactor Trips consists of alert and bistable indication lights EIIS:IL! which will light and sometimes flash when a channel in the 7300 Process Control System (PCS) receives a signal that a setpoint has been reached (e.g. low coolant flow). When logic is satisfied (e.g. two out of three channels in a loop receive a low coolant flow signal) a Reactor Trip is initiated and a first out annunciator that indicates what caused the Reactor Trip (e.g. Lo Flow P-8 Permissive Reactor Trip) will light.

The P-8 permissive interlock EIIS:IEL! blocks a Reactor Trip from either a single loop loss of coolant flow signal or a Turbine EIIS:TRB! Trip signal if the unit is below approximately 48 percent of full power.

The Reactor Coolant (NC) System EIIS:AB! low flow Reactor Trips protect the core from departure from nucleate boiling (DNB) in the event of a loss of coolant flow. An output signal from two out of the three bistables in a loop indicates a low flow in that loop. Above P-8 permissive, low flow in any one loop causes a Reactor Trip. Each of the two logic trains, A and B, is capable of opening a separate and independent Reactor Trip breaker, RTA and RTB, respectively.

Technical Specification Action Statement No. 14, Table 3.3-3 states:

"With the number of operable channels one less than the minimum channels operable requirement, be in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is operable."

Description of Event

On August 26, 1989 at 0934, a Reactor Trip signal was received along with a "Lo Flow P-8 Permissive Reactor Trip" first out alarm indication. The reactor was in Mode 1, Power Operation, at 100 percent power with all control systems in automatic prior to the trip. There were no NC low flow alert or bistable indications

received prior to receiving the Reactor Trip and the first out Reactor Trip indication. Instrumentation and Electrical (IAE) personnel were testing the RPS 7300 PCS System in Channel IV with T-ave and delta-T in test. The Events Recorder EIIS:IQ! had documented that the Reactor Trip Breaker A (RTA) opened 284 milliseconds prior to the opening of Reactor Trip breaker B (RTB).

The Reactor Trip initiated a Turbine Trip. The C Steam Generator (S/G) EIIS:SG! Power Operated Relief Valve (PORV) EIIS:RV), 1SV-7, lifted as the S/G pressure reached 1119.3 psig which is within the open setpoint range (1114 psig to 1136 psig). Next, S/G B had a low-low level signal which in turn initiated the start of the Auxiliary Feedwater (CA) System EIIS:BA!.

Operations (OPS) Control Room personnel implemented Reactor Trip procedure, EP/1/A/5000/01, Reactor Trip or Safety Injection, and then entered procedure EP/1/A/5000/1.3, Reactor Trip. OPS Control Room personnel then implemented Reactor Trip Recovery procedure OP/1/A/6100/05, Unit Fast Recovery. At 0950, a manual isolation of the Feedwater (CF) System EIIS: SJ! was initiated to conserve auxiliary steam. At about 1000 the NRC was notified by OPS Control Room personnel according to procedure RP/0/A/5700/10, NRC Immediate Notification

Requirements.

At 1030, Work Request 139505 was submitted to check calibration on the NC system flow bistables for the RPS. Extensive testing was performed to determine if any of the bistable setpoints had drifted above the required trip setpoint (90 percent). After testing was complete, it was determined that the bistables had not drifted. The flows for all loops were plotted and none of the flows had decreased below the bistable setpoints. This indicated that the trip did not occur because of an input to the 7300 PCS channels.

A meeting was then held between Operations, IAE, Maintenance Engineering Services (MES) and Performance personnel at 1800 to discuss the current findings from troubleshooting the 7300 PCS cabinet EIIS: CAB). Former trip reports were reviewed for documented times for Reactor Trip Breakers to trip. A loss of voltage trip is normally somewhere between 60 to 90 milliseconds with an administrative maximum of 100 milliseconds. Based on past Reactor Trips, it was shown that it took about 190 milliseconds to initiate a Reactor Trip on a high negative reactivity rate (i.e. all control rods entering the core by gravity). IAE tested the 7300 PCS output signals to the annunciators since no alert or bistable low flow indications were received prior to the trip. Normally, if one channel receives a low flow signal, an alert annunciator will alarm. No problems were found. Therefore, with the lengthy time between Reactor Trip breakers and correct annunciator operation, MES personnel concluded that the problem was probably in the SSPS Train "A" cabinet.

At 1820, Work Request 139510 was submitted to troubleshoot the SSPS Train "A" Cabinet. Train "A" was chosen because RTA opened first on a loss of voltage signal. RTB opened on a high negative reactivity rate signal.

Procedures IP/0/A/3010/07, Procedure For Troubleshooting The Solid State Protection System (SSPS), and PT/0/A/4601/08A, Solid State Protection System (SSPS) Train A Periodic Test With NC System Pressure Greater Than 1955 psig, were used to troubleshoot the SSPS Train "A" cabinet. Universal boards EIIS:ECBD! A303, 304,

305, 306, and 308 were tested with the SSPS Cabinets' automatic logic test sequencer. This sequencer sends pulses to the circuit boards to check logic state changes. No problems were found. The boards were checked again by manually checking the logic states and no problems were found. A visual check of the boards was completed by pulling the boards out of the cabinet, thereby, deenergizing the boards. By procedure when the boards are removed, they are required to have the logic re-checked.

When Universal board A303 was checked this time, it failed. All the other boards tested satisfactorily. IAE then tested board A303 again with it failing again. A303 was replaced. Also, by procedure, when a board is replaced the entire cabinet is required to undergo a logic check. All boards checked out satisfactorily.

At 2218, OPS complied with Technical Specification Action Statement No. 14 of Table 3.3-3 since SSPS Train "A" was not returned to service within the required 2 hours because of troubleshooting. Unit 1 was in Mode 3, Hot Standby, at the time. At 2220, Universal Board A303 in the SSPS Train "A" cabinet was determined to have failed and was replaced. At 2230, a follow-up meeting between Operations, IAE, MES, and Performance personnel was held to inform these groups about the failed universal board. At 2240, SSPS Train "A" was returned to service after testing had been completed.

On August 28, 1989, at 1230, Unit 1 was returned to Mode 1, Power Operation.

Conclusion

This event is assigned a cause of Equipment Failure/Malfunction because Universal Board A303 failed. When Universal Board A303 failed, an immediate signal was sent to Universal Board A308 confirming a low NC flow signal. This then requited an output signal to be sent to trip the Reactor if the P-8 permissive interlock was unblocked. Since the unit was at 100 percent power, the signal was sent to RTA to trip the Reactor. While the Reactor was shutting down, RTB tripped on high negative power rate.

Since 1986, there have been 9 failures out of 156 Universal boards in use at the station. This is a 2.0 percent failure rate per year. None of the past Universal board failures have caused a Reactor Trip. The past failed Universal boards were sent to Westinghouse to be repaired. The Universal board is manufactured by Westinghouse Electric Corporation, model number 6056D21 G01 and serial number 1206.

MES personnel have contacted Westinghouse personnel about repairing this Universal board. Work Request No. 69501 has been written to send the-Universal-board to Westinghouse for repair. A repair summary has been requested to document what caused the board to fail.

One anomaly resulting from the Reactor Trip was that S/G B level decreased to 11 percent, Station personnel are continuing to investigate the cause of this level decrease. The CA system automatically started because of the S/G 13 low-low level signal and was capable of removing the decay heat from the Reactor.

A review of the LER data base for the past twelve months prior to this event revealed 2 LERs that described Reactor Trips resulting from equipment failures. LER 370-89-02 documented a Reactor Trip because of a S/G B Low-Low level following loss of the CF turbine pump (CFPT) 2B. This was a failure of suction pressure switches on the CFPT and a broken air supply line to the Full Load Rejection valve EIIS:V! 2CM-420. LER 370/89-03 documented a Reactor Trip because of a low-low level in S/G C which was caused by a failed bellows EIIS: BE! in the positioner for the CF regulating valve for S/G C, 2CF-20. The corrective actions for these LERs were specific to their events. These events are not similar because the failures were on different equipment. Therefore, this event is considered not recurring.

This event is Nuclear Plant Reliability Data System (NPRDS) reportable. Industry reported to NPRDS seven Universal board failures for boards used in the RPS which were manufactured by Westinghouse Electric Corporation with the model number 6056D21 G01. Four board failures were caused by faulty integrated circuit chips and three failures were caused by unknown reasons.

There were no personnel injuries, radiation overexposures, or releases of radioactive material as a result of this event.

CORRECTIVE ACTIONS:

Immediate: 1) OPS personnel implemented procedure EP/1/A/5000/01, Reactor Trip or Safety Injection, and then entered procedure EP/1/A/5000/1.3, Reactor Trip.

2) OPS personnel implemented recovery procedure OP/1/A/6100/05, Unit Fast Recovery.

Subsequent: IAE personnel replaced Universal Board A303 in the SSPS train "A" cabinet.

Planned: 1) MES personnel will send the Universal board to Westinghouse for repair and failure analysis.

2) The McGuire Safety Review Group will write an addendum to this LER when the repair summary is received from Westinghouse.

SAFETY ANALYSIS:

An analysis of a decrease in NC System flow rate is presented in Section 15.3, Decrease in Reactor Coolant System Flow Rate, of the Final Safety Analysis Report (FSAR). If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly. The necessary protection against a partial loss of coolant flow

accident is provided by the low primary coolant flow Reactor Trip signal which is actuated in any reactor coolant loop by two out of three low flow signals. Above Permissive 8, low flow in any loop will actuate a Reactor trip. The analysis of effects and consequences for low NC flow shows that the DNB ratio will not decrease below the limit value at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

The unit responded to the Reactor Trip without any significant problems. All primary and secondary system parameters were at their approximate no-load value 30 minutes after the trip.

As Main Steam EIIS: SB! pressure for S/G C reached 1119.3 psig, PORV 1SV-7 lifted because the S/G pressure entered the open setpoint range (1114 psig to 1136 psig) for valve 1SV-7. S/G A, B, and D PORVs did not actuate, although S/G A, B, and D pressures entered the open setpoint range. None of the S/G pressures exceeded their respective open setpoint ranges. Main Steam pressure did not reach the main steam code safety valve lift setpoints and the valves were not challenged. NC system pressure did not reach the Pressurizer EIIS:PZR! PORV or Pressurizer Code Safety valve lift setpoints and the valves were not challenged. Adequate core cooling was maintained throughout this transient, and the NC System boundary was not challenged. Emergency power and emergency core cooling were not required in this event and were not actuated.

The health and safety of the public were not affected by this incident.

ADDITIONAL INFORMATION:

Sequence Of Events

OAC - Operator Aid Computer Printout

TRI - Transient/Reactor Trip Investigation Report
SSL - Unit 1 Shift Supervisor's Logbook
SEL - Unit 1 Shift Engineer's Logbook
REL - Unit 1 Reactor Engineer's Logbook.
PR - Personnel Recollection
WR - Work Request Section 1

Date Time Event

8/26/89 09:33:54 A Reactor Trip was received on a first out alarm "Lo Flow P-8 Permissive Reactor Trip." (OAC,TRI,SEL,SSL)

09:34:01 A Reactor Trip initiated a Turbine Trip. (OAC,TRI)

09:34:17 S/G C PORV 1SV-7 lifted as S/G pressure reached 1119.3 psig; thereby, entering the Open Setpoint Range for valve lSV-7. (OAC,TRI)

09:34:22 S/G B Low-Low Level signal caused the CA system to start. (TRI)

09:50:10 A manual feedwater isolation was initiated to conserve auxiliary steam. (TRI)

---- OPS Control Room personnel implemented the Reactor Trip recovery procedures. (TRI,PR)

approx. OPS Control Room personnel made the required 1000 notification to the NRC. (SEL,PR)

1030 Work Request No. 139505 was submitted to check calibration on the NC system flow bistables for RPS. (WR)

1800 - A meeting was held to discuss the results of the 1930 troubleshooting findings and what further testing should be performed. (PR)

1820 Work Request 139510 was submitted to troubleshoot the SSPS Train "A" Cabinet. (WR)

2218 SSPS Train "A" was not returned to service within two hours of commencing testing; therefore, OPS complied with Technical Specification Action Statement No. 14. (SSL)

2220 The Universal Board in the SSPS Train "A" cabinet was found failed and was replaced. (PR,SEL)

2230 A follow-up meeting was held to inform station groups of the failed board. (PR,SEL)

2240 SSPS Train "A" was back in-service with testing completed. (SSL)

8/28/89 1230 Unit 1 was retur
ed to Mode 1, Power Operation.

SUPPLEMENTAL INFORMATION:

The following information addresses Planned Corrective Action Number 2, to submit the results of the failure summary analysis performed by Westinghouse on Universal Board A303.

MES personnel sent Universal Board A303 to Westinghouse. On February 12, 1990, MES personnel received a Repair Report from Westinghouse on the item. Westinghouse reported that after the universal board was received, it was inspected on January 31, 1990 with no visible damage or defects noted. The universal board was then sent for testing on February 5, 1990. During this test, two component failures were noted. The component failures were on CR29 and CR23 diodes. The CR29 diode is only associated with demultiplexing and could not have contributed to the Reactor Trip. However, the CR23 diode is tied directly to an output signal which generates the Reactor Trip signal by de-energizing the Reactor Trip Breaker Undervoltage Coil. Consequently, the failure of the CF23 diode caused the Reactor to trip. To cause a Reactor Trip, CF23 diode would have to have been electrically shorted out. It could not be determined how CF23 became shorted.

ATTACHMENT 1 TO 9006110089 PAGE 1 OF 2

Duke Power Company (704)875-4000
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

DUKE POWER

May 30, 1990

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: McGuire Nuclear Station Unit 1
Docket No. 50-369
Licensee Event Report 369/89-22-01

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 369/89-22-01 concerning a Reactor Trip because of a failed universal board in the Solid State Protection System. This report is being revised and submitted in accordance with 10 CFR 50.73(a)(2)(iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

T. L. McConnell

DVE/ADJ/cbl

Attachment

xc: Mr. S. D. Ebneter American Nuclear Insurers
Administrator, Region II c/o Dottie Sherman, ANI Library
U. S. Nuclear Regulatory The Exchange, Suit 245
Commission 270 Farmington Avenue
101 Marietta St., NW, Suite 2900 Farmington, CT 06032
Atlanta, GA 30323

INPO Records Center Mr. Darl Hood
Suite 1500 U. S. Nuclear Regulatory
1100 Circle 75 Parkway Commission
Atlanta, GA 30339 Office of Nuclear Reactor
Regulation
M&M Nuclear Consultants Washington, D. C. 20555
1221 Avenue of the Americas
New York, NY 10020 Mr. P. K. Van Doorn
NRC Resident Inspector
McGuire Nuclear Station

LER 369/89-22-01

bxc: B. W. Bline
L. G. Bost
J. S. Warren
R. L. Gill
R. M. Glover (CNS)
T. D. Curtis (ONS)
P. R. Herran
S. S. Kilborn (W)
R. E. Lopez-Ibanez
M. A. Mullen
R. O. Sharpe (MNS)
G. B. Swindlehurst
K. D. Thomas
M. S. Tuckman
L. E. Weaver
R. L. Weber
J. D. Wylie (PSD)
J. W. Willis
QA Tech. Services NRC coordinator (EC 12/55)
MC-815-04
(20)

*** END OF DOCUMENT ***